UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

December 21, 2006

William R. Brian, Acting Vice
President, Operations
Grand Gulf Nuclear Station
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: INSPECTION REPORT 050-00416/06-014; 072-00050/06-003

Dear Mr. Brian,

This report covers four visits made by the NRC to your Grand Gulf Station between October 9 and November 18, 2006. The purpose of the site visits was to inspect your dry fuel storage preoperational testing activities and to observe your initial fuel loading operation. The spent fuel cask crane and crane support structure were evaluated on October 9-13, 2006. Integration of dry fuel storage operations into your station 10 CFR Part 50 programs was evaluated on October 16-20, 2006. The heavy loads segment of your ISFSI pre-operational testing program was observed on October 23 through November 3, 2006. And initial loading of spent fuel into dry storage was observed on November 10-18, 2006. The enclosed inspection report presents the results of the inspection, which were discussed with members of your staff at the conclusion of each site visit.

The inspection determined that you have completed dry fuel storage pre-operational testing and loaded your first canister of spent fuel into dry storage in accordance with the Commission's rule and regulations and within the condition of your license. No violations were identified.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Should you have any questions concerning this inspection, please contact the undersigned at (817) 860-8191 or Mr. Scott Atwater at (817) 860-8286.

Sincerely,

D. Blair Spitzberg, Ph.D., Chief
Fuel Cycle and Decommissioning Branch

Docket Nos.: 50-416 / 72-050
License No.: NPF-29

Enclosure: NRC Inspection Report
050-00416/06-014; 072-00050/06-003

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	050-00416; 072-00050
License No.:	NPF-29
Report No:	050-00416/06-014; 072-00050/06-003
Licensee:	Entergy Operations, Inc.
Facility:	Independent Spent Fuel Storage Installation Grand Gulf Nuclear Station Entergy Operations, Inc. P.O. Box 756 Port Gibson, MS 39150
Dates:	October 9 through November 18, 2006
Inspectors:	S.P. Atwater, Health Physicist, Inspection Lead R.L. Kellar, P.E., Health Physicist L.M. Willoughby, Resident Inspector Fort Calhoun Station
Accompanied By:	R.B. Reeves, NMSS/DHLWRS A.P. Sapountzis, NMSS/DHLWRS M.G. Karmis, NMSS/DSFST
Approved By:	D.B. Spitzberg, Ph.D., Chief Fuel Cycle and Decommissioning Branch
Attachments:	1. Supplemental Information 2. Inspector Notes

EXECUTIVE SUMMARY

Grand Gulf Nuclear Station
NRC Inspection Report 050-00458/05-014; 072-00049/05-003

Entergy had selected the Holtec HI-STORM 100 system for dry storage of spent nuclear fuel at the Grand Gulf Station. The Nuclear Regulatory Commission (NRC) had certified the Holtec cask system for storage of irradiated fuel under Certificate of Compliance (CoC) No. 72-1014.

Between October 9 and November 18, 2006, the NRC conducted an inspection of the Grand Gulf spent fuel dry storage activities. The spent fuel cask crane and crane support structure were evaluated on October 9-13, 2006, as part of the heavy loads program. The evaluation included a review of the crane's inspection, testing and maintenance history and review of the crane's licensing basis documents. Grand Gulf's integration of dry fuel storage operations into the station 10 CFR Part 50 programs and incorporation of the dry fuel storage Technical Specifications into the procedures was evaluated on October 16-20, 2006. The heavy loads segment of the Grand Gulf pre-operational testing program was observed on October 23 through November 3, 2006, to evaluate the preparedness of the personnel and adequacy of procedures to load fuel into dry storage. Initial loading of spent fuel into dry storage was observed on November 10-18, 2006, without incident and within the limits of the Technical Specifications.

Prior to the site visit on October 9-13, 2006, the spent fuel cask crane had not been fully tested, inspected or maintained in accordance with the requirements of American Society of Mechanical Engineers (ASME) B30.2. The licensee reported that the crane had been used infrequently and had not been operated at or near rated capacity. Prior to handling spent fuel, the crane was inspected, refurbished and tested to bring it into compliance with ASME B30.2 requirements. Therefore, no violations were identified with the use of the crane to move spent fuel under 10 CFR Part 72 requirements. However, the inspectors identified several potential issues associated with crane use and maintenance as required by 10 CFR Part 50, which are described in Attachment 2. The potential issues are planned to be evaluated by Region IV Division of Reactor Safety (DRS) during the Permanent Plant Modifications, 10 CFR 50.59 and Maintenance Rule inspections scheduled during 2007.

Details related to the activities observed during this inspection are provided in Attachment 2 to this report. The following is a summary of the major activities observed.

- The crane and crane support structure were evaluated for performance during a Safe Shutdown Earthquake (SSE) while sustaining a 150 ton load. The analysis determined that the crane controlled the load and remained on the tracks, and that the auxiliary building walls maintained their structural integrity (Attachment 2, Crane Design & Licensing).

- Crane inspections identified significant materiel deficiencies. These deficiencies were corrected prior to dry fuel loading operations. Cold-proof testing and performance testing of the crane had not been performed in accordance with the ASME B30.2 and Regulatory Guide 1.104. Prior to handling spent fuel, the crane was repaired and tested to bring it into compliance with the ASME B30.2 requirements (Attachment 2, Crane Inspection, Load Testing and Performance Tests).

- ISFSI operations had been adequately incorporated into the Grand Gulf Emergency Plan. Controls had been established for revising the emergency plan, consistent with regulatory requirements (Attachment 2, Emergency Planning).

- ISFSI operations had been adequately incorporated into the Grand Gulf Fire Plan. The fire and explosion hazards analyses performed for the ISFSI pad and cask transport path were complete and comprehensive (Attachment 2, Fire Protection).

- The spent fuel assemblies selected for the first dry fuel loading campaign met Technical Specification requirements. A post loading verification was performed to verify proper loading of fuel into the canister and the Special Nuclear Material (SNM) accountability program tracked the transfer of spent fuel assemblies from the spent fuel pool to the ISFSI (Attachment 2, Fuel Selection/Verification).

- The evaluation report required by 10 CFR 72.212 discussed Grand Gulf's compliance with requirements established in the Holtec Certificate of Compliance (CoC), Final Safety Analysis Report (FSAR) and in 10 CFR Part 50 (Attachment 2, General License).

- Evaluations performed pursuant to 10 CFR Parts 50.59 and 72.48 for heavy lifts had been completed as required by License Condition 5. Safe load paths had been defined and a minimum transfer cask lifting height had been established as required by NUREG 0612 (Attachment 2, Heavy Loads).

- Written procedures had been developed for dry fuel storage operations as required by License Condition 2. The procedures were consistent with the technical basis described in the Final Safety Analysis Report (FSAR) and contained the Technical Specification requirements (Attachment 2, Procedures and Tech Specs).

- The licensee had applied their NRC approved 10 CFR Part 50 Quality Assurance Program to the construction and operation of the ISFSI. The condition reports reviewed indicated that conditions adverse to quality were being promptly identified, documented, reported to the appropriate levels of management and corrected in a timely manner. Material receipt inspection checklists had been developed and non-conforming materials, parts and components were segregated, isolated and tagged to preclude inadvertent installation or use (Attachment 2, Quality Assurance).

- The ALARA (As Low As Reasonably Achievable) measures required for maintaining personnel exposures within the limits of 10 CFR 20 were established and implemented during pre-operational testing and initial fuel loading (Attachment 2, Radiation Protection).

- Radiation levels from a fully loaded ISFSI pad were calculated and used to establish site-specific dose rate limits for the loaded storage cask and transfer cask. These limits ensured that an individual located on or beyond the site boundary would receive less than the regulatory limits during design basis accidents and during normal operation and anticipated occurrences (Attachment 2, Radiation Protection).

- A safety review program satisfying the requirements of 10 CFR 72.48 had been applied to dry fuel storage operations. The criteria for determining if a change, test or experiment required a license amendment was consistent with the criteria specified in 10 CFR 72.48 (Attachment 2, Safety Reviews).

- An NRC approved training program was used for training personnel in dry fuel storage operations. It included system design and licensing bases, operations, radiation protection procedures, and expected radiation dose rates. The pre-operational testing activities required by License Condition 10 were successfully completed prior to initial loading (Attachment 2, Training).

ATTACHMENT 1

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel:

C. Abbot - Quality Audits Supervisor
R. Benson - Radiation Protection Supervisor
R. Capps - Fuel Services Mechanic
J. Clements - Radiation Protection Technician
A. Forte - Radiation Protection Technician
D. Hall - Fuel Services Mechanic
D. Ellis - Fuel Services Senior Project Manager
E. Harris - Nuclear Safety Assurance
R. Ingram - Operations Senior Reactor Operator
E. Jones - Administrative Services
J. Jones - Radiation Protection Technician
OB Magee - Dry Fuel Services Supervisor
R. Martin - Reactor Engineer
H. Neely - Fuel Services Mechanic
H. Nelson - Fuel Services Electrician
J. Owens - Senior Licensing Specialist
T. Pate - Fuel Services Electrician
S. Raner - Fuel Services Mechanic
L. Robertson - Fuel Services Manager
D. Townsend - Senior Emergency Planner
B. Warren - Engineering Senior Project Manager
T. Worthington - Engineering Programs Supervisor

INSPECTION PROCEDURES USED

60854	Preoperational Testing of an Independent Spent Fuel Storage Installation
60855	Operation of an Independent Spent Fuel Storage Installation

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

None

Closed

None

Discussed

None

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BTU	British Thermal Unit
BWR	Boiling Water Reactor
CARB	Corrective Action Review Board
CDR	Cask Document Record
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DFS	Dry Fuel Storage
ER	Engineering Request
FHA	Fire Hazards Analysis
FHD	Forced Helium Dehydration
FSAR	Final Safety Analysis Report
GWD/MTU	Gigawatt Days per Metric Ton Uranium
ICA	Item Control Area
ISFSI	Independent Spent Fuel Storage Facility
ISG	Interim Staff Guidance
kW	Kilowatt
MWD/MTU	Megawatt Days per Metric Ton Uranium
M&TE	Measuring and Test Equipment
MPC	Multi-Purpose Canister
NDE	Non-Destructive Examination
OBE	Operating Basis Earthquake
OJT	On-The-Job Training
PIC	Person-In-Charge
PM	Preventive Maintenance
QA	Quality Assurance
RIR	Receiving Inspection Report
NRC	U.S. Nuclear Regulatory Commission
RVOA	Removable Valve Operating Assembly
RWP	Radiation Work Permit
SAT	Systematic Approach to Training
SCS	Supplemental Cooling System
SER	Safety Evaluation Report
SNM	Special Nuclear Material
SSC	System, Structure, or Component
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
TER	Technical Evaluation Report
UFSAR	Updated Final Safety Analysis Report
VCT	Vertical Cask Transporter
wt.%	Weight percent

Attachment 2

Grand Gulf ISFSI Pre-Operational Testing and Initial Loading

(Inspector Notes - Table of Contents)

Attachment 2

Grand Gulf ISFSI Pre-Operational Testing and Initial Loading

(Inspector Notes)

Category: Crane Design & Licensing **Topic:** NUREG 0612 Phase I & II Letters

Reference: GL 81-07, GL 85-11

Requirement: NRC Generic Letter 81-07 was issued on December 22, 1980. The Generic Letter requested nuclear power plant licensees to review their controls for handling heavy loads and to to submit a report to the NRC documenting the results of their review. The specific information requested by the NRC was identifed in Sections 2.1 through 2.4 of Enclosure 3 to the letter, which described how the licensee intended to meet the guidelines contained in NUREG-0612. The licensees were requested to submit the information identified in Section 2.1 within 6 months (Phase I) and the information identified in Sections 2.2, 2.3 and 2.4 wihin 9 months (Phase II).

Finding: This requirement was implemented. Mississippi Power and Light submitted their Grand Gulf Phase I report to the NRC on November 23, 1981 and their Phase II report on May 4, 1982. On May 24, 1982, the NRC contractor (EG&G Idaho) completed a draft Technical Evaluation Report (TER) for Grand Gulf's Phase I and Phase II reports. In Supplement 5 to the Grand Gulf NRC Safety Evaluation Report (SER) dated August 1984, the NRC concurred with the NRC Contractor TER for the Grand Gulf Phase I and Phase II reports. In a letter from the NRC to Mississippi Power and Light dated April 4, 1985, the NRC concluded that Grand Gulf had met the guidelines of NUREG 0612 and that no further action was required.

Documents Reviewed:
Letter from Mississippi Power and Light to the NRC dated November 23, 1981
Letter from Mississippi Power and Light to the NRC dated May 4, 1982
EG&G Idaho draft Technical Evaluation Report (TER) dated May 24, 1982 for Grand Gulf's Phase I and Phase II reports
Supplement 5 to the NRC Safety Evaluation for Grand Gulf, dated August 1984
Letter from the NRC to Mississippi Power and Light dated April 4, 1985.

Category: Crane Design & Licensing **Topic:** Seismic Load Control Criteria

Reference: NUREG 0554, Section 2.5

Requirement: Overhead cranes may be operating at a time that an earthquake occurs. The crane should be designed to retain control of and hold the load, and the bridge and trolley should be designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event.

Finding: This requirement was implemented. The Safe Shutdown Earthquake (SSE) seismic analysis was performed assuming the crane was loaded to 150 tons. The analysis was performed for the trolley at mid-span and at the end of travel, and for the hook at the full down and full up positions. The peak lateral and vertical forces (includng pendulum swing) during an SSE occurred with the trolley at mid-span with the load full up. The dominant mode was the vertical forces and uplift lugs were required in order to keep the truck wheels on the trolley rails during an SSE.

Documents Reviewed: Grand Gulf Calculation #C-H035.0, "Refueling Crane - 150 Ton Capacity", Revision 0

Category: Crane Design & Licensing **Topic:** Support Structure

Reference: GG UFSAR, Sections 3.8.4.1.1.1; 3.8.6.2; 3.8.6.3

Requirement: The auxiliary building is designed to maintain its structural integrity under normal operating conditions, Operating Basis Earthquakes (OBE), Safe Shutdown Earthquakes (SSE) and tornados. The 150 ton spent fuel cask crane is supported by reinforced concrete walls which transmit the load to the foundation through the exterior wall and structural steel columns imbedded in the concrete.

Finding: This requirement was implemented. The maximum lateral and vertical forces exerted on the bridge rails during an SSE with the crane loaded to 150 tons were determined through the seismic analysis for the crane. These forces were then applied to the seismic analysis for the auxiliary building walls. The maximum lateral and vertical forces transmitted to each wall joint, horizontal reinforcement, and structural steel column during an SSE were bounded by the wall and foundation design.

Documents Reviewed: Calculation #9645, "Exterior Walls - Crane Loads - Auxiliary Building", dated May 14, 1976
Calculation #C-H035.0, "Refueling Crane - 150 Ton Capacity", Revision 0

Category: Crane Design & Licensing **Topic:** Wire Rope Breaking Strength

Reference: ASME B30.2, Section 2-1.14.3(a)

Requirement: The hoisting ropes shall be of a recommended construction for crane service. The total rated load divided by the number of parts of line shall not exceed 20% of the minimum breaking strength of the rope. (This is the same as a 5 to 1 safety margin.)

Finding: This requirement was implemented. Grand Gulf used a Whiting 150 ton single-failure-proof crane with two independent and fully redundant wire ropes. Each wire rope contained 3 sheaves and 6 parts of line. For this configuration, the minimum required breaking strength for each wire rope was 250,000 lbs. During destructive testing performed by Universal Wire Products, Inc. on January 26, 1977, a specimen from each wire rope was tested. Both specimens parted at a tensile stress in excess of 250,000 lbs.

Documents Reviewed: Universal Wire Products, Inc., Certificate of Test dated January 26, 1977

Category: Crane Inspection **Topic:** Annual Inspections

Reference: ASME B30.2, Section 2-2.1.3

Requirement: Periodic or annual crane inspections shall be performed to check: (a) deformed, cracked or corroded members, (b) loose or missing bolts, nuts, pins or rivets, (c) cracked or worn sheaves and drums, (d) worn, cracked or distorted parts, (e) excessive wear of brake system, (f) excessive wear of drive chain sprockets and excessive drive chain stretch, (g) deterioration of controllers or switches, (h) motion limit devices that interrupt power, and (i) rope reeving.

Finding: This requirement was implemented. The inspection requirements for periodic crane inspections were located in the various procedures listed below. A periodic inspection was performed on the Spent Fuel Cask Crane on August 12, 2006 under Work Orders 83456 and 83933 using the applicable procedures. The following deficiencies were identified at that time:

The bridge and trolley travel limit switches were found to be inoperable. Troubleshooting determined that the switches were not repairable and replacement switches were no longer available. The licensee evaluated the loss of the limits switches, as allowed by ASME B30.2, and determined that the loss did not constitute a hazard. The licensee was developing a design change for replacement of the switches.

The bridge south girder seismic storage lock was found to be inoperable. The locking pin and its Tee Handle operating tool were repaired on October 23, 2006 under Work Order 60595-01.

The bridge and trolley hydraulic brakes were found to be marginally operable. The brakes were inspected, cleaned, adjusted, and retested satisfactorily on October 23, 2006 under Work Order 93979-08.

An incorrect viscosity lubricating oil was found in the bridge drive unit and hoist motor gearbox. The oil analyses showed high levels of particulate, wear metals and moisture. Oil was drained from the gear-motor and drive unit gearboxes on the bridge, trolley and hoist on September 17, 2006 under Work Order 93689-1. All gear boxes were refilled with manufacturer recommended lubricating oil and returned to service.

The hoist primary bull gear and pinion gears were found to be dry and galled. Condition Report 2006-03131 was generated to clean, inspect, and evaluate the condition of the gears. The galling was determined to be surface scuffing and minor abrasive wear on the contact surfaces caused by inadequate lubrication. Gear tooth structural integrity was not challenged. The gears were re-lubricated and returned to service.

The hoist creep (inching) drive was found to be inoperable. The creep drive was repaired on September 23, 2006 under Work Order 93979-01 and returned to service.

The wire rope equalizing system was found to be inoperable. The system required a minimum operating pressure of 500 psig. The as-found system pressure was 125 psig and hydraulic fluid was not present in either shock suppressor. The wire rope equalizing system was repaired on October 22, 2006, under Work Order 93979-07. The accumulator was initially charged to 250 psig with nitrogen. Both hydraulic shock suppressors were then filled with hydraulic fluid and hydraulic pressure was raised to 514 psig. No hydraulic fluid leakage was subsequently identified and this requirement was fully implemented.

Maintenance issues and crane deficiencies discovered during this inspection will be evaluated by RIV Division of Reactor Safety (DRS) during a future inspection.

Documents Reviewed: Procedure 07-S-05-300, "Control and Use of Cranes and Hoists", Revision 111

Procedure 07-S-14-228, "Frequent PM Checks Spent Fuel Cask Crane", Revision 3
Procedure 07-S-05-330, "Control and Use of Crane and Hoist Rigging", Revision 11
Procedure 07-S-15-2, "Frequent Inspection of Cranes", Revision 11
Procedure 07-S-14-226, "Spent Fuel Cask Crane Periodic Inspection", Revision 6
Work Order 60595-01, "Seismic Storage Pin Broken", dated October 20, 2006
Work Order 93979-08, "Inspect & Adjust Brakes", dated October 23, 2006
Condition Report 2006-03131, "Evaluate Drum Gear Abrasion", dated October 16, 2006
Work Order 93979-01, "Crane - Recent PMs", dated September 21, 2006
Work Order 93979-07, "Inspect and Rework Hydraulic System", dated October 22, 2006
Work Order 93689-01, "Drain Oil, Flush and Replenish", dated September 15, 2006

Category: Crane Inspection **Topic:** Hook Inspections

Reference: ASME B30.10, Section 10-1.4

Requirement: Hooks shall be inspected for: a) any bend or twist exceeding 10% from the plane of an unbent hook; b) a throat opening greater than 15% of the original opening for hooks without latches or 8% for hooks with latches; c) wear exceeding 10% of the original section dimension; d) cracks, severe nicks or gouges; and e) inoperative latch. These inspections should be performed monthly during normal service, weekly to monthly during heavy service and daily to weekly during severe service.

Finding: This requirement was implemented. Procedure 07-S-15-2, Section 7.2 performed a visual inspection of the hook monthly, or prior to use if the crane had been out of service for greater than one month. The inspection criteria was consistent with the ASME code.

Documents Reviewed: Procedure 07-S-15-2, "Frequent Inspection of Cranes", Revision 11

Category: Crane Inspection **Topic:** Wire Rope Inspections

Reference: ASME B30.2, Sections 2-2.4.1(a) and 2-2.4.2

Requirement: All wire ropes should be visually inspected at the start of each shift. A thorough inspection of all wire ropes shall be made at least once a month and a certification record, which includes the date of inspection, the signature of the person who performed the inspection and an identifier for the ropes which were inspected shall be kept on file. The annual inspection should include the entire length of the wire rope. The wire rope shall be replaced when inspection finds: a) reduced rope diameter below nominal; b) worn or broken outside wires; c) corroded or broken wires at end connections; d) corroded, cracked, bent, worn or improperly applied end connections; and e) severe kinking, crushing, cutting or unstranding.

Finding: Some of this requirement had not been implemented at the time of the crane inspection on October 9, 2006. Procedure 07-S-05-300, Attachment II contained the crane daily and shiftly checks that must be performed prior to each use. Steps 1.6 and 1.8 of the Attachment checked the hoist wire rope and end fittings for excessive wear, broken wires, stretch kinking, twisting, and proper lay on all sheaves and the hoist drum. This wire rope inspection was observed during both pre-operational testing and initial loading.

Procedure 07-S-15-2, Section 7.3 visually inspected the wire rope monthly, or prior to use if the crane had been out of service for greater than one month. The inspection checked for a reduction in wire rope diameter below nominal, worn or broken outside wires, corroded or broken wires at end connections, corroded, cracked, bent, worn or improperly applied end connections, and severe kinking, cutting or unstranding.

However, documentation that a full length wire rope inspection had been performed within the past year was not located. A full length wire rope inspection was conducted on October 17, 2006 under Work Order 93979-05. The load block was lowered to the 133' elevation in the auxiliary building, which was the lowest elevation to which the cask crane load block would travel during dry fuel storage operations. At the 133' elevation, 17 wraps of wire rope remained on the hoist drum. The deployed length of wire rope and the 3 wire rope clamp bolts on the drum were inspected using criteria consistent with the ASME code. No deficiencies were identified. This inspection satisfied the referenced requirement.

Maintenance issues and crane deficiencies discovered during this inspection will be evaluated by RIV Division of Reactor Safety (DRS) during a future inspection.

Documents Reviewed: Procedure 07-S-05-300, "Control and Use of Cranes and Hoists", Revision 111
Procedure 07-S-15-2, "Frequent Inspection of Cranes", Revision 11
Work Order 93979-05, "Inspect Main Hoist Cables and 3 Clamp Bolts on Drum", dated October 17, 2006

Category: Crane Load Testing **Topic:** Cold-Proof Testing

Reference: Regulatory Guide 1.104, Section C.4.d

Requirement: A cold-proof test at 125 percent of rated capacity should be performed once very 40 months to verify the absence of brittle fracture tendencies in the ferritic load carrying members of the crane. Following cold-proof testing, non-destructive examination of the welds whose failure could result in the drop of a critical load should be performed.

Finding: This requirement had not been fully implemented at the time of the crane inspection on October 9, 2006. The initial cold-proof testing had been conducted under Bechtel Work Plan Q1T31-Y-10208Y02 on September 24, 1980. The test was conducted using test weights totaling 374,000 pounds rather than the minimum required 375,000 pounds. The Grand Gulf UFSAR stated that the minimum expected operating temperature for the crane was 60 degrees F. The temperature at which the original cold-proof testing was performed had not been documented in the testing package and therefore a minimum operating temperature for the crane could not be established.

This requirement was fully implemented on October 30, 2006 following a repeat of cold-proof testing under Work Order 96746-01. The total test weight was 397,860 pounds (198.93 tons). The sequence included raising and lowering the load to heights between 6 and 12 inches with interim holding periods to verify the brakes could hold the load with no slippage. The trolley was then moved east and west and the bridge was moved north and south approximately 3 feet to test the bridge and trolley brakes.

The Grand Gulf spent fuel cask crane was initially licensed under Regulatory Guide 1.104, which later became NUREG-0554. The NRC approved Grand Gulf's use of Regulatory Guide 1.104 for cold-proof testing and Grand Gulf elected to remain under those requirements. Regulatory Guide 1.104, Section C.4.d allowed increasing the test load beyond the crane design rated capacity in order to achieve a reduction in minimum operating temperature. Every 1.5 percent increase in test load above the rated capacity yielded a one degree reduction in minimum operating temperature. Grand Gulf elected to increase the test load by 11.25 tons (7.5 percent of 150 tons)) in order to achieve a minimum operating temperature of five degrees F below the cold-proof testing temperature. Cold-proof testing was conducted at approximately 198.75 tons (187.5 + 11.25). The average ambient temperature at the crane upper structure during the load test was 66.4 degrees F. Therefore, a minimum temperature of 61.5 degrees F was established for operating the crane. Procedure 07-S-05-335, Step 6.4.11 contained this temperature limit.

Regulatory Guide 1.104, Section C.4.d required the cold-proof testing to be followed by non-destructive examination (NDE) of the critical welds. The Grand Gulf UFSAR, Appendix 3A/1.104, Subject (4) eliminated the NDE requirement since the structural members and joints were designed with a minimum safety factor of five based on the ultimate strength. The licensing position stated that the higher design factor of safety, in conjunction with cold-proof testing, was sufficient to demonstrate the structural integrity of the crane, thus non-destructive testing was not necessary.

Crane deficiencies discovered during this inspection will be evaluated by RIV Division of Reactor Safety (DRS) during a future inspection.

Documents Reviewed: Bechtel Work Plan Q1T31-Y-10208Y02, "Load Test of Auxiliary Building Spent Fuel Cask Crane", issued on August 5, 1980.
Grand Gulf UFSAR, Appendix 3A/1.104-1, Subject (4)
Work Order 96746-01, "Perform Cold Proof Load Test of the Cask Handling Crane", dated October 27, 2006
Procedure 07-S-05-335, "Operation of the Spent Fuel Cask Crane", Revision 4
.

Category: Crane Load Testing **Topic:** Hook Load Testing

Reference: NUREG 0554, Section 4.3

Requirement: A 200% static type load test should be performed for each load-attaching hook. Measurements of the geometric configuration of the hooks should be made before and after the test and should be followed by a nondestructive examination that should consist of volumetric and surface examinations to verify the soundness of fabrication and ensure integrity of the hooks. The results of examinations should be documented and recorded.

Finding: This requirement was implemented. The 150 ton Whiting crane main hook eye and sister hook were individually load tested to 600,000 pounds by Talbot laboratories on January 25, 1977. Dimensional measurements were taken before and after the testing and no evidence of inelastic behavior was identified. Magnetic particle testing of the main hook eye and sister hook was performed by the Whiting Corporation after the load testing and no relevant indications were identified.

Documents Reviewed: Talbot Laboratory Report for the Proof Load Test of the 150 ton Sister Hook, dated February 1, 1977

Talbot Laboratory Report for the Proof Load Test of the 150 ton Eye, dated February 1, 1977
Whiting Corporation Magnetic Particle Report, dated February 18, 1977

Category: Crane Performance Tests **Topic:** Governing Code

Reference: ASME B30.2, Section 2-2.2.1

Requirement: Prior to initial use, all new, reinstalled, altered, extensively repaired or modified cranes shall be tested to ensure that the hoist, bridge and trolley will travel correctly and that all limit switches, locking devices and safety devices are operable.

Finding: This requirement had not been fully implemented at the time of the crane inspection on October 9, 2006. The Bechtel Work Plan required a dynamic performance test under rated load in which the crane was exercised through all motions at rated speeds. The performance test was conducted on September 25, 1980 using a 150 ton test weight set. The test load was raised to a sufficient height to clear all obstacles and the bridge and trolley were fully traveled in both directions. The bridge and trolley brakes were tested at full speed in both directions. The hoist was traveled through full upward and downward motions. However, the Bechtel performance testing on September 25, 1980 did not test: a) actuation of the electric holding brakes on hoist motor overspeed and on loss of phase; b) engagement of the mechanical load brakes on loss of the electric holding brakes; and c) operation of the alternator equipped eddy-current brake on loss of all brakes. Condition report 2006-03977 was generated to document NRC identification of these omissions and the crane was removed from service pending resolution.

Prior to initial fuel loading, this requirement had been fully implemented. The hoist high speed motor and low speed motor overspeed devices were tested on October 27, 2006 under Work Order 93979-06. Both as-found trip setpoints were off-scale high and the electric holding brakes would not have actuated on hoist motor overspeed. The high speed motor overspeed device was reset to 750 rpm and the low speed motor overspeed device was reset to 16 rpm. These setpoints corresponded to 125 percent of rated speed.

The hoist motor loss of phase feature was tested on October 21, 2006 under Work Order 93979-06. The phase loss relay was opened to simulate a loss of phase and the electric holding brakes actuated as designed.

Both mechanical load brakes were independently tested on September 23, 2006 and again on September 26, 2006 under Work Order 93979-01. In both tests, the north load brake failed to stop and hold the load. The brake adjusting nut was discovered frozen and the brake could not be adjusted. The licensee evaluated the loss of the north mechanical load brake under Engineering Request ER-GG-2005-009-001, ERCN-004 and concluded that the loss did not adversely affect the reliability and functionality of the crane. Since the hoist was equipped with dual, independent gear trains, either mechanical load brake was capable of holding the load on failure of the other. The south load brake and the eddy-current brake provided adequate protection against load drop on failure of both electric holding brakes.

The eddy-current brake was tested on October 20, 2006 under Work Order 93979-06. The voltage and amperage readings across the brake alternator were taken during hoist lowering and found to be within specifications.

Maintenance issues and crane deficiencies discovered during this inspection will be evaluated by RIV Division of Reactor Safety (DRS) during a future inspection.

Documents Reviewed: Bechtel Work Plan Q1T31-Y-10208Y02, "Load Test of Auxiliary Building Spent Fuel Cask Crane", issued on August 5, 1980
Condition Report 2006-03977, "NRC Notification of Potential Crane Issues", dated October 14, 2006
Work Order 93979-06, "Perform 9 Specific Inspection/Test", dated October 20, 2006
Work Order 93979-01, "Crane - Recent PMs", dated September 23, 2006
Engineering Request ER-GG-2005-009-001, ERCN-004, "Spent Fuel Cask Handling Crane", Revision 0

Category: Crane Performance Tests **Topic:** Load Hangup Protection

Reference: NUREG 0554, Section 4.5

Requirement: As an alternative to designing the crane to absorb or control the kinetic energy of the rotating machinery during load hangup, the system design may include a load cell system in the drive train, a motor current-sensing device, or a mechanical load-limiting device that will de-energize the hoist drive motor and the main power supply under a load hangup condition.

Finding: This requirement had not been implemented at the time of the crane inspection on October 9, 2006. The cask handling crane was equipped with a load cell in the drive train and an over-current relay in the hoist motor circuitry. These two devices were designed to independently de-energize the hoist drive motor under a load hangup condition.

During initial testing on September 25, 1980 the hoist load cell circuit was inoperable. During the crane inspection on August 12, 2006 the load cell circuit was found disabled (leads lifted). No evidence could be found that the load cell had ever been operable. During initial testing on September 25, 1980, the hoist motor over-current relay was not tested. The relay was tested on October 21, 2006 under Work Order 93979-06 and found to be set to greater than 200 amps. The over-current relay would not have protected the crane against a load hangup condition until load exceeded 200 tons. At the time of the crane inspection on October 9, 2006, the spent fuel cask crane had no load hangup protection.

This requirement was implemented prior to initial fuel loading. The hoist motor over-current relay was reset to 130 amps, which was the equivalent to 125 percent of the full rated load. Troubleshooting of the load cell circuit under Work Order 86218 indicated the system was not repairable. The load cell system was no longer manufactured and parts were no longer available. While the licensee was developing a design change for a replacement load cell system, the hoist motor over-current circuit was providing load hangup protection.

Maintenance issues and crane deficiencies discovered during this inspection will be evaluated by RIV Division of Reactor Safety (DRS) during a future inspection.

Documents Reviewed: Bechtel Work Plan Q1T31-Y-10208Y02, "Load Test of Auxiliary Building Spent Fuel Cask Crane", issued on August 5, 1980
Work Order 93979-06, "Perform 9 Specific Test/Inspection", dated October 20, 2006

Category: Crane Performance Tests **Topic:** Stop Feature

Reference: NUREG 0554, Sections 3.3, 6.1 and 6.6

Requirement: An emergency stop feature should be installed at the control station. For cranes remotely operated using radio control stations, a second emergency stop feature should be provided at ground level to remove power from the crane, independent of the controller. Cranes that use more than one control station should be provided with electrical interlocks that permit only one control station to be operated at a time.

Finding: This requirement had not been implemented at the time of the crane inspection on October 9, 2006. The emergency stop feature was not tested during the initial testing on September 25, 1980 and evidence of a subsequent test was not identified.

The Grand Gulf cask handling crane was equipped with two STOP pushbuttons, one in the cab and the other on the pendant. A remote radio control feature was not provided. A two-position switch was provided in the cab for selecting either cab or pendant control of the crane. Depressing either STOP button would interrupt power to the crane and would apply the bridge, hoist and trolley brakes. Pendant control was not used during dry cask storage operations. The STOP pushbutton in the crane cab was tested on October 21, 2006 under Work Order 93979-06 and it performed as designed. This requirement was implemented.

Maintenance issues and crane deficiencies discovered during this inspection will be evaluated by RIV Division of Reactor Safety (DRS) during a future inspection.

Documents Reviewed: Work Order 93979-06, "Perform 9 Specific Test/Inspection", dated October 20, 2006

Category: Crane Performance Tests **Topic:** Two-Block Protection

Reference: NUREG 0554, Section 4.5

Requirement: As an alternative to designing the crane to absorb or control the kinetic energy of the rotating machinery during a two-blocking condition, a system of upper travel limit switches may be used to prevent two-blocking. The system should include two independent travel limit devices of different designs and activated by separate mechanical means. These devices should de-energize the hoist drive motor and the main power supply.

Finding: This requirement was implemented. The Grand Gulf cask handling crane hoist was equipped with two upward travel limit switches. The primary switch was actuated by a geared rotary device and the backup switch was actuated when the load block contacted a mechanical bar. These two devices were designed to independently de-energize the hoist drive motor to prevent two-blocking.

Both limit switches were tested on October 16, 2006 under Work Order 93979-04. The primary switch was disabled and the hoist was raised until the backup switch de-enegized the hoist motor. The hoist was then lowered and the primary switch was returned to service. The hoist was again raised until the primary switch de-enegized the hoist motor.

Documents Reviewed: Work Order 93979-04, "Check/Test the 2 block Limit", dated October 16, 2006

Category: Emergency Planning **Topic:** Emergency Plan

Reference: 10 CFR 72.32(c)

Requirement: Each ISFSI must have an Emergency Plan. For an ISFSI that is located on the site of a nuclear power plant licensed for operation, the site emergency plan required by 10 CFR 50.47 shall be deemed to satisfy this requirement.

Finding: This requirement was implemented. The ISFSI had been incorporated into the Grand Gulf Emergency Plan. Procedure 10-S-01-1 required activation of the Emergency Plan for any accident involving a breach of the dry fuel storage canister confinement boundary.

Procedure 10-S-01-26, Section 6.4 required offsite emergency response personnel to receive annual training. The 2006 Annual Offsite Agency Training was conducted on June 20-21, 2006 and included site layout and access (including the ISFSI pad), basic health physics and radiation protection and fire protection measures. Representatives from the state and local emergency response organizations attended the training. Specific training in the care of radiation accident victims was provided to local emergency response organization and hospital personnel during 2005 and 2006.

The offsite emergency response agencies completed their 2005 annual review of the emergency preparedness letters of agreement with Grand Gulf on December 19, 2005 with no requests for corrections or changes.

Procedure 01-S-10-4 tasked the Manager, Emergency Preparedness with defining the scope, scheduling and conduct of drills and exercises. Attachment II to Procedure 01-S-10-4 required annual drills involving radiological monitoring, emergency preparedness, medical emergency and fire. Drills to date had not involved the ISFSI since it was not operational. Condition Report WT-GGN-2006-00000 was developed for ensuring the ISFSI was incorporated into future drills.

Documents Reviewed:
Procedure 10-S-01-1, "Activation of the Emergency Plan", Revision 115
Procedure 10-S-01-26, "Offsite Emergency Response", Revision 10
Procedure 01-S-10-4, "Emergency Preparedness Drills and Exercises", Revision 11
Grand Gulf letter 2006/00399 dated October 12, 2006. Vicksburg Fire Department MS-1 Training
Grand Gulf letter 2005/00490 dated September 14, 2005. MS-1 Hospital Training for Claiborne County Hospital
Grand Gulf letter 2005-00632 dated December 5, 2005. MS-1 Hospital Training for Riverland Medical Center
Grand Gulf letter 2006-00003 dated January 3, 2006. MS-1 Hospital Training for River

Region Medical Center
Grand Gulf letter 2006/00398 dated October 12, 2006. MS-1 Training for Emergystat Ambulance Service Personnel
Grand Gulf letter 2006/00246 dated June 22, 2006. 2006 Annual Offsite Agency Training
Grand Gulf letter 2006/00004 dated January 3, 2006. Emergency Preparedness Letter of Agreement (LOA) Annual Review - 2005
Condition Report WT-GGN-2006-00000, "EP Emergency Plan", dated November 8, 2006

Category: Emergency Planning **Topic:** Emergency Plan Revisions

Reference: 10 CFR 72.44(f)

Requirement: The licensee may make changes to the emergency plan without prior NRC approval provided the changes do not decrease the effectiveness of the plan. Within six months of any changes made to the emergency plan, the licensee shall submit a report containing a description of the changes to the appropriate regional office and to the Spent Fuel Project Office.

Finding: This requirement was implemented. Procedure 01-S-10-3, Steps 2.4.10 and 6.4.7 required the licensee to distribute revisions to the Emergency Plan and its implementing procedures to the NRC within 30 days of implementation.

Documents Reviewed: Procedure 01-S-10-3, "Emergency Preparedness Department Responsibilities", Rev. 13

Category: Fire Protection **Topic:** Fire Accident Response

Reference: FSAR 1014, Sect 11.2.4.4

Requirement: Upon detection of a fire adjacent to a loaded HI-TRAC or HI-STORM overpack, the ISFSI operator shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC as the pressure relief valves may have opened and water loss from the water jacket may have occurred resulting in an increase in radiation dose. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

Finding: This requirement was implemented. Procedure 10-S-03-2 established the actions to be taken by the fire brigade upon discovery of a fire. A radiation protection technician was assigned to each fire brigade team. The radiation protection technician was responsible for ensuring the team took the appropriate radiological precautions. Procedure 20-S-01-003 required a post-fire visual inspection of the cask for damage and a radiation survey for loss of shielding effectiveness.

Documents Reviewed: Procedure 10-S-03-2, "Response to Fires", Revision 18
Procedure 20-S-01-003, "DFS HI-STORM / HI-TRAC Transport", Revision 0

Category: Fire Protection **Topic:** Fire and Explosion Hazards Analysis

Reference: CoC 1014, TS B.3.4.5; FSAR 1014, Sect 2.2.3.3

Requirement: The potential for fire or explosion shall be addressed, based on site specific considerations. This includes the condition that the onsite transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded storage cask or transfer cask.

Finding: This requirement was implemented. Engineering Request ER-GG-2003-0018-003 analyzed the fire and explosion hazards in the auxiliary building, on the ISFSI pad, and along the transport route. The purpose of the licensee's Fire Hazards Analysis (FHA) was to demonstrate that any potential fire would be bounded by the Holtec design basis fire. The licensee' evaluation considered the heat loading from all combustible materials that would be present during each phase of dry fuel loading operations, the effective fire zone area, and the proximity of the fire to the transfer cask and storage cask.

The Holtec design basis fire had a combustible loading of 49,673 British Thermal Unit (BTU) per square foot and a peak fire temperature of 1,475 degrees F. The Fire Hazards Analysis showed that potential fires that could be encountered during dry fuel storage loading activities had a maximum combustible loading of 16,993 BTU per square foot and a peak temperature of 1399 degrees F. Therefore, all potential fires were bounded by the Holtec design basis fire.

Engineering Request ER-GG-2003-0018-003 also established the following restrictions along the transport route during movement of the storage cask from the auxiliary building to the ISFSI pad: a) no fuel, compressed gas or liquified compressed gas deliveries; b) no vehicle parking within 50' of the transport route; c) the Vertical Cask Transporter (VCT) and hyster forklift must not be in the vicinity of the storage cask at the same time; d) no testing or intentional running of telecom diesel generator; and e) stop all work in the combination shop when the crawler was within 300' of the building. Attachment I of Procedure 20-S-01-003 implemented these restrictions.

Documents Reviewed: ER-GG-2003-0018-003, "Fire Hazard Evaluation for the Grand Gulf ISFSI Cask Hauling and Storage", Revision 1
Procedure 20-S-01-003, "DFS HI-STORM / HI-TRAC Transport", Revision 0

Category: <u>Fire Protection</u> **Topic:** <u>Offsite Emergency Support</u>

Reference: 10 CFR 72.122(g)

Requirement: Systems, structures and components (SSCs) important to safety must be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance services, and other emergency agencies.

Finding: This requirement was implemented. The Grand Gulf ISFSI and transport route were provided with ready access for emergency vehicles and personnel. Procedure 10-S-01-26, Section 6.5 established the methods for escorting emergency response vehicles and personnel onto the site and to the scene.

Documents Reviewed: Procedure 10-S-01-26, "Offsite Emergency Response", Revision 10

Category: <u>Fuel Selection/Verification</u> **Topic:** <u>Acceptable Contents for Storage; MPC-68</u>

Reference: CoC 1014, TS B.2.1.1; Tables 2.1-1, 2.1-3, 2.1-8

Requirement: The MPC-68 may be loaded with uranium oxide or mixed oxide fuel, intact or damaged fuel assemblies, and non-fuel hardware. Fuel debris is not authorized. The fuel assemblies must meet the criteria for maximum planar-average initial enrichment, initial maximum rod enrichment, cooling time, average burnup, and decay heat. The non-fuel hardware must meet the criteria for cooling time and average burnup.

Finding: This requirement was implemented. The first Grand Gulf loading campaign consisted of four casks, each containing 68 intact Boiling Water Reactor (BWR) fuel assemblies of the 8X8C, 9X9A, 9X9E and 9X9F array and class. Non-fuel hardware, fuel debris, damaged fuel assemblies, fuel assemblies with stainless steel clad or channels, and fuel assemblies with high burnup were not loaded in the first campaign. At the time of the inspection, the licensee had identified 272 fuel assemblies, plus 14 alternates, as candidates for loading into the first four casks. Of these, the 68 fuel assemblies needed for the first cask had been selected. The individual characteristics of each fuel assembly were contained in the licensee's fuel assembly database named "Cask Loader".

Technical Specification Table 2.1-3 limited the maximum planar-average initial enrichment to 4.2 wt.% U-235 for the 8X8C and 9X9A fuel assemblies, and to 4.0 weight percent (wt.%) U-235 for the 9X9E and 9X9F fuel assemblies. Table 5-2 of the Cask Loader database indicated the spent fuel assemblies identified as candidates for the first cask had a maximum planar-average initial enrichment of 3.98 wt.% U-235.

Technical Specification Table 2.1-3 limited initial maximum rod enrichment to 5.0 wt.% U-235. Table 5-2 of the Cask Loader database indicated the spent fuel assemblies identified as candidates for the first cask had a maximum rod enrichment of 4.90 wt.% U-235.

Maximum allowable fuel assembly average burnup varied with cooling time, decay heat, and minimum fuel assembly average enrichment. Technical specification B.2.4.3.2 provided an equation for integrating these variables to reach a value for maximum allowable fuel assembly average burnup. This equation had been incorporated into the licensee's Cask Loader computer program. The results of the Cask Loader calculations were presented in Attachment V of Procedure 17-S-02-111. The maximum allowable fuel assembly average burnup was 51,139 MWD/MTU and the maximum burnup of any assembly selected for the first cask was 35,881 MWD/MTU.

Table 4.4.21 of the Holtec FSAR limited the MPC-68 canister total heat load to 28.19 kilowatts (kW). Attachment V of Procedure 17-S-02-111 indicated the total decay heat load for the first canister was 12.608 kW.

Documents Reviewed: Procedure 17-S-02-111, "Fuel Selection for Dry Storage," Revision 0
Calculation NEAD-SR-06/001, "GGNS Cycles 1-10 Cask Loader Database," Revision 0

Category: Fuel Selection/Verification **Topic:** Damaged Fuel Classification

Reference: FSAR 1014, Table 1.0.1; ISG-1, Rev. 2

Requirement: A damaged fuel assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel locations that are not replaced with dummy fuel rods, or one that cannot be handled by normal means. Interim Staff Guidance (ISG) 1, "Damaged Fuel", Revision 2 provides more detailed information.

Finding: This requirement was implemented. Attachment I of Procedure 17-S-02-111 defined damaged fuel in a manner consistent with the Holtec FSAR and Interim Staff Guidance, ISG -1. The licensee had classified the fuel assembly candidates for the first loading campaign as intact based on a combination of reactor operating records, chemistry data, and fuel assembly visual inspection results.

Documents Reviewed: Procedure 17-S-02-111, "Fuel Selection for Dry Storage," Revision 0
Interim Staff Guidance, ISG -1, "Damaged Fuel," Revision 2

Category: Fuel Selection/Verification **Topic:** Post Loading Verification

Reference: FSAR 1014, Section 8.1.4.3

Requirement: Perform a post-loading visual verification of the assembly identification markings to confirm that the serial numbers match the approved fuel loading pattern.

Finding: This requirement was implemented. The post-loading visual verification was performed in accordance with Procedure 20-S-01-007 and Attachment I. Each spent fuel assembly serial number was independently inspected and verified to have been loaded into the cell location specified by the loading plan contained in Attachment VI of Procedure 17-S-02-111. The post loading verification was videotaped and no anomalies were identified.

Documents Reviewed: Procedure 20-S-01-007, "DFS Fuel Movement and Position Verification", Revision 0
Procedure 17-S-02-111, "Fuel Selection for Dry Storage," Revision 0

Category: Fuel Selection/Verification **Topic:** Special Nuclear Material Inventory and Records

Reference: 10 CFR 72.72(a)

Requirement: Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, and transfer of all Special Nuclear Material (SNM) with quantities specified in 10 CFR 74.13(a)(1).

Finding: This requirement was implemented. The inventory, transfer and storage of Special Nuclear Material (SNM) at Grand Gulf was controlled through Procedure EN-NF-200. Attachment 9.1 to Procedure EN-NF-200 identified the reactor core, spent fuel pool and ISFSI as Item Control Areas (ICAs) for storage of SNM. Procedure EN-NF-200, Section 5.2 required that movement of all SNM between ICAs be controlled by an ICA Transfer Form.

Procedure 20-S-01-003, Step 6.4.49 required the Refueling Supervisor in Charge (RSIC) to complete an SNM transfer sheet once the storage cask was placed on the ISFSI pad. The SNM transfer sheets were provided in Attachment VIII to Procedure 17-S-02-300.

Documents Reviewed: Procedure EN-NF-200, "Special Nuclear Material Control", Revision 2
Procedure 20-S-01-003, "DFS HI-STORM / HI-TRAC Transport" Revision 0
Procedure 17-S-02-300, "Fuel and Core Component Movement Control", Revision 117
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Category: General License **Topic:** 72.212 Report - Compliance With CoC

Reference: 10 CFR 72.212(b)(2)(i)(A)

Requirement: A general licensee shall perform written evaluations, prior to use, that establish that the conditions set forth in the Certificate of Compliance (CoC) have been met.

Finding: This requirement was implemented. Section IV of the 10 C FR 72.212 evaluation report provided details of how each of the applicable Certificate of Compliance requirements had been met.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: General License **Topic:** 72.212 Report - Compliance With FSAR

Reference: 10 CFR 72.212(b)(3)

Requirement: The general licensee shall review the Final Safety Analysis Report (FSAR) and NRC Safety Evaluation Report, prior to use of the general license, to determine whether the reactor site parameters, including analysis of earthquake intensity and tornado missiles, are enveloped by the cask design basis. The results of this review must be documented in the 10 CFR 72.212(b) Evaluation Report.

Finding: This requirement had not been fully implemented at the time of the programs inspection on October 16, 2006. The licensee had adequately evaluated all of the reactor site parameters and had incorporated them into the 10 CFR 72.212(b) evaluation report, with the exception of snow loading, lightning and cask burial under debris. These remaining reactor site parameters had been evaluated and incorporated into the 10 CFR 72.212(b) evaluation report prior to initial fuel loading and this requirement was implemented.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5
Holtec 1014 FSAR, Revision 4, Section 2.2, "HI-STORM Design Criteria,"

Category: General License **Topic:** 72.212 Report - Compliance With Part 50

Reference: 10 CFR 72.212(b)(4)

Requirement: Prior to use of the general license, determine whether activities related to storage of spent fuel involve a change in the facility technical specifications or require a license amendment for the facility pursuant to 10 CFR Part 50.59(c)(2). Results of this determination must be documented in the 10 CFR 72.212(b) Evaluation Report.

Finding: This requirement was implemented. Section C.3.11 of the 10 CFR 72.212 evaluation report documented that several 10 CFR 50.59 safety reviews had been made by the licensee for design modifications associated with implementation of dry fuel storage, and that none of the modifications required a change to the Grand Gulf Nuclear Station operating license or technical specifications. The 50.59 screening for the 10 CFR 72.212 evaluation stated that changes to the facility specifications were not required.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5
50.59 Screening for ER-GG-2003-0018-000, "ISFSI Implementation, Operation and Compliance to Requirements of Generic Holtec FSAR and CoC and Incorporation of 212 Report for GGNS," Revision 0

Category: General License **Topic:** Analysis - Blocked Air Vents on the Storage Cask

Reference: CoC 1014, Tech Spec B.3.4.9

Requirement: For those users whose site specific design basis includes an event that results in blockage of the storage cask inlet or outlet air vents for greater than 24 hours, an analysis may be performed to demonstrate adequate heat removal for the duration of the event. If the analysis is not performed or adequate heat removal cannot be verified, alternate methods of cooling must be established.

Finding: This requirement was not applicable at the Grand Gulf ISFSI. There were no postulated site-specific design basis events (mudslides, floods, etc.) that could potentially result in the blockage of any HI-STORM inlet or outlet ducts for an extended period of time. The maximum flood from the Mississippi River including backwater effects from drainage ditches would not reach the height of the bottom of the HI-STORM cask located on the ISFSI pad.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: General License **Topic:** Analysis - Cask Design Basis Drop

Reference: CoC 1014, Tech Spec B.3.4.6.a

Requirement: For free standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to less than or equal to 45 g's at the top of the canister fuel basket. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with American National Standards Institute (ANSI) N14.6 and has redundant drop protection features.

Finding: This requirement was implemented. The licensee had elected to meet ISFSI pad parameter Set A, consisting of a concrete thickness of less than or equal to 36 inches in thickness, a concrete compressive strength of less than or equal to 4,200 psi at 28 days, and a subgrade effective modulus of less than or equal to 28,000. The licensee concrete thickness and compressive strength and subgrade effective modulus met all the parameter Set A requirements.

The "4-Point Lift Systems" Vertical Cask Transporter (VCT) was designed in accordance with ANSI N14.6 as a special lifting device, however the redundant drop protection feature was not used by the licensee. Procedure 20-S-01-003, Step 6.3.1.a therefore limited the maximum storage cask lift height to 10.5 inches above the ISFSI pad surface. During pre-operational testing and initial fuel loading, the storage cask lift height was controlled to less than 6 inches above the surface.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: General License **Topic:** Analysis - Cask Spacing

Reference: FSAR 1014, Section 1.4, Tables 1.4.1 and 1.4.2

Requirement: Generic spacing criteria are provided in Table 1.4.1 for two by N arrays and in Table 1.4.2 for large arrays. The casks should be arrayed in such a manner that the tributary area for each cask (open ISFSI area attributable to a cask) is a minimum of 225 square feet. For specific sites, a smaller tributary area of cask spacing requirements can be used after appropriate thermal evaluations are performed.

Finding: This requirement was implemented. Appendix C, Section C.5.1 of the 10 CFR 72.212 Evaluation Report stated that the Grand Gulf ISFSI used a cask spacing of 15 feet by 16 feet and was considered a square array. The tributary area was the sum of the orthogonal pitches (15 X 16), or 240 square feet. This exceeded the minimum required tributary area of 225 square feet specified in the Holtec FSAR.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5
Letter from Holtec to Grand Gulf, dated October 19, 2006

Category: General License **Topic:** Analysis - Flood Conditions

Reference: CoC 1014, Tech Spec B.3.4.4

Requirement: The analyzed flood condition of 15 feet per second (fps) water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.

Finding: This requirement was implemented. Appendix C of the 72.212 report determined that the elevation of the water for the probable maximum flood from the Mississippi River, including backwater effects from drainage ditches, would not exceed 132 feet 11 inches mean sea level. The bottom of the loaded HI-STORM casks located on the ISFSI pad were at an elevation of 133 feet, 0 inches mean sea level. Therefore, the HI-STORM casks would not be exposed to running water of 15 fps with a height of 125 feet.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: General License **Topic:** Analysis - Limiting Site Temperatures

Reference: CoC 1014, Tech Spec B.3.4.1; 3.4.2

Requirement: The maximum average yearly temperature is 80 degrees F. The temperature extremes, averaged over a 3-day period, shall be greater than -40 degrees F and less than 125 degrees F.

Finding: This requirement was implemented. The average maximum average yearly temperature at Grand Gulf was 75 degrees F, and the record maximum and minimum daily extreme temperatures were listed as 104 degrees F and -1 degree F in the Grand Gulf UFSAR.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: General License **Topic:** Analysis - Seismic Acceleration of Casks

Reference: CoC 1014, Tech Spec B.3.4.3

Requirement: Free-standing storage casks may be deployed at sites when it can be shown that design basis earthquake acceleration values are below the threshold needed to cause cask tip-over or excessive sliding, when the coefficient of friction between the cask and the ISFSI pad is 0.53 or greater. For sites with environmental conditions that may degrade the coefficient of friction, an analysis should be performed to demonstrate that a design basis earthquake will not result in cask tip-over or cause a cask to slide off the pad. Impacts between casks should be precluded, or shown to result in fuel deceleration values of 45 g's or less. For sites with design basis earthquake acceleration values higher than those allowed for free-standing casks, the casks shall be anchored to the ISFSI pad.

Finding: This requirement was implemented. Section C.4.3.2.3 of the 72.212 report documented that the casks as implemented at Grand Gulf would not tip-over or slide excessively during a seismic event. The licensee analyzed the HI-STORM overpack and ISFSI pad to determine that the equation provided in Certificate of Compliance Appendix B, Section 3.4.3 was satisfied. The licensee also evaluated cask sliding considering icy conditions and determined that the amount of displacement would be less than the spacing between the casks on the ISFSI pad.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: General License **Topic:** Revisions to the 72.212 Report

Reference: 10 CFR 72.212(b)(2)(ii)

Requirement: The general licensee shall evaluate any changes to the 10 CFR 72.212 report in accordance with the requirements of 10 CFR 72.48(c). A copy of this record shall be retained until spent fuel is no longer stored under the general license.

Finding: This requirement had not been implemented at the time of the programs inspection on October 16, 2006. Procedure EN-LI-112 provided two categories for retention of the 10 CFR 72.48 evaluations. The first category specified retention of the record for the life of the license for evaluations of changes to the facility. The second category specified a retention period of five years for evaluations of changes to procedures and tests or experiments. A 72.48 evaluation supporting the 72.212 report would fall into the second category.

Condition Report 2006-04172 was generated to clarify the intent and was assigned to Entergy Corporate Licensing. The next revision to Procedure EN-LI-112 was scheduled for March 1, 2006, at which time this requirement will be implemented.

Documents Reviewed: Procedure EN-LI-112, "10 CFR 72.48 Review Program", Revision 2
Condition Report 2006-04172, "Revise EN-LI-112 to reflect the requirements of 10 CFR 72.212(b)(2) C (ii)", dated October 25, 2006

Category: Heavy Loads **Topic:** Licensed Facility Heavy Loads Requirements

Reference: CoC 1014, Condition 5

Requirement: Each lift of a canister, transfer cask or storage cask must be made in accordance with the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant specific review (under 50.59 or 72.48, if applicable) is required to show operational compliance with existing plant specific heavy loads requirements.

Finding: This requirement was implemented. Engineering Request ER-GG-2003-0018 evaluated each heavy lift to be made during dry fuel storage operations. Each evaluation considered the potential impact of a load drop on safety related equipment, as well as the adequacy of the slings, crane design, lifting procedures and crane operator training. The evaluation concluded that the dry fuel storage heavy lifts could be made within the existing Grand Gulf heavy loads requirements and procedures.

The 50.59 and 72.48 screenings for ER-GG-2003-0018 concluded that the dry fuel storage heavy lifts did not adversely affect a design function, a method of performing or controlling a design function, or a method of evaluation of a design function of a System, Component, or Structure (SSC) as described in the UFSAR.

Documents Reviewed: Engineering Request ER-GG-2003-0018, "Evaluation of Heavy Lifts for Compliance with NUREG 0612 and GGNS-CS-20", Revision 0
50.59 and 72.48 Review Forms for Engineering Request ER-GG-2003-0018, dated October 30, 2006

Category: Heavy Loads **Topic:** Safe Load Paths

Reference: NUREG 0612, Section 5.1.1 (1)

Requirement: Safe load paths should be defined for the movement of heavy loads to minimize the potential for a drop of a heavy load to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled.

Finding: This requirement was implemented. Procedure 20-S-01-002, Step 6.3.2.g required all heavy loads inside the auxiliary building to be moved via their safe load path. The dry fuel loading safe load path was defined with laser targets stenciled on the floor of the auxiliary building at the 208' elevation. Targets were provided for transfer cask movements between the cask loading pool, cask washdown pit, and train bay. A laser was installed on the underside of the crane trolley and was directed downward onto the targets. For each move, the Person-In-Charge (PIC) directed the crane operator to align the crane such that the laser was within the target. The laser targets were accurately positioned during initial fuel loading.

Documents Reviewed: Procedure 20-S-01-002, "DFS Cask Loading", Revision 0

Category: Heavy Loads **Topic:** Synthetic Roundsling Inspection Prior to Use

Reference: ASME B30.9, Sections 9-6.9.2 and 9-6.9.4

Requirement: Synthetic Roundslings shall be inspected for damage each day or shift the sling is used. A synthetic round sling shall be removed from service if any of the following conditions are present: a) missing or illegible sling identification; b) acid or caustic burns; c) heat damage; d) holes, tears, cuts abrasive wear or snags that expose the core yarns; e) broken or damaged core yarns; f) weld splatter that exposes core yarns; g) round slings that are knotted; h) discoloration and brittle or stiff areas which may mean chemical or ultraviolet/sunlight damage; or i) fittings that are pitted, corroded, cracked, bent, twisted, gouged or broken.

Finding: This requirement was implemented. Attachment 20 to Procedure 20-S-01-005 contained the inspection requirements for the synthetic roundslings used on the dry fuel storage project. The inspection points and removal from service criteria were consistent with the ASME code. The slings were consistently inspected prior to each use during both pre-operational testing and initial fuel loading.

Documents Reviewed: Procedure 20-S-01-005, "DFS Rigging Plan", Revision 0

Category: Heavy Loads **Topic:** Transfer Cask Minimum Lift Height

Reference: Crane Manufacturer Specification

Requirement: For single failure proof cranes, the transfer cask height during movement should be sufficiently high to allow for engaging of the brakes during an uncontrolled descent.

Finding: This requirement was implemented. Engineering Request ER-GG-2003-0018 stated that, based on correspondence with Whiting, the maximum distance the load would drop on loss of both electric holding brakes was 8 inches when the load was at its lowest travel on the 133 foot elevation. Procedure 20-S-01-002, Step 6.3.1 established a minimum lift height of 9 inches above the floor and this minimum lift height was reiterated before each heavy load movement within the loading sequence. The minimum lifting height was consistently established during pre-operational testing and initial fuel loading.

Documents Reviewed: Engineering Request ER-GG-2003-0018, "Evaluation of Heavy Lifts for Compliance with NUREG 0612 and GGNS-CS-20", Revision 0
Procedure 20-S-01-002, "DFS Cask Loading", Revision 0

Category: Procedures & Tech Specs **Topic:** Operating Procedures

Reference: CoC 1014, Condition 2

Requirement: Written operating procedures shall be prepared for cask handling, loading, and movement. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the FSAR.

Finding: This requirement was implemented. The first dry fuel storage campaign at Grand Gulf was conducted using written and approved procedures that were consistent with the technical basis described in Chapter 8 of the Holtec #1014 FSAR. The campaign was conducted under Work Order #68259 and Radiation Work Permit #2006-1081. The procedures used to conduct the initial dry fuel storage campaign are listed below. All procedures were strictly followed.

Documents Reviewed: Work Order #68259
Radiation Work Permit #2006-1081, "DFS Cask Load and Transport", Revision 0
Procedure 20-S-01-002, "Dry Fuel Cask Loading", Revision 0
Procedure 20-S-01-003, "DFS HI-STORM / HI-TRAC Transport", Revision 0
Procedure 20-S-01-005, "DFS Rigging Plan", Revision 0
Procedure 20-S-01-006, "DFS Radiological Monitoring Requirements For The HI-STORM 100", Revision 0
Procedure 20-S-01-007, "DFS Fuel Movement and Position Verification", Revision 0
Procedure 20-S-01-015, "DFS Vertical Cask Transporter Operation", Revision 0
Procedure 20-S-01-140, "DFS MPC Forced Helium Drying Operations", Revision 0
Procedure 04-1-01-F11-3, "Fuel Handling Platform", Revision 32

Category: Procedures & Tech Specs **Topic:** Storage Cask Heat Transfer Validation

Reference: CoC 1014, Condition 9

Requirement: The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first cask system of each design placed in service by any user, with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR. Validation tests shall be performed for each subsequent cask system that has a heat load exceeding the previous cask by 2 kW up to 16 kW. A letter summarizing each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4.

Finding: This requirement was implemented. Grand Gulf referenced the thermal validation test performed at Energy Northwest's Columbia Generating Station in Engineering Request ER-GG-2003-0018-000, Section C.4.1.2. Columbia loaded Cask #120 on March 16, 2004. The cask contained an MPC-68 with a decay heat load of 17.1 kW and the thermal behavior was as predicted. Columbia Generating Station submitted the test results to the NRC on July 28, 2004. The first cask placed in service at the Grand Gulf station contained a decay heat load of 12.608 kW.

Documents Reviewed: ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5
Holtec letter #9042868 to Energy Northwest, dated July 12, 2004.
Energy Northwest letter #GO2-04-134 to the NRC, dated July 28, 2004 (ML0421903320).

Category: Procedures & Tech Specs **Topic:** Transfer Cask Lifting Trunnion Inspection

Reference: FSAR 1014, Table 9.2.1

Requirement: The transfer cask lifting trunnions must undergo annual load testing or dimensional testing.

Finding: This requirement was implemented. The transfer cask trunnions and trunnion pads were inspected by River Bend Station under Work Order 65552-01 prior to shipping the transfer cask to Grand Gulf. The trunnions underwent a dimensional check and a visual examination for scratches, indentation, surface wear and distortion. The trunnions and trunnion pads were subjected to Magnetic Particle Testing. No deficiencies were identified.

Documents Reviewed: Work Order 65552-01, "HI-TRAC Trunnion Inspection", dated 7/20/06

Category: Quality Assurance **Topic:** Approved Quality Assurance Program

Reference: 10 CFR 72.140(d)

Requirement: A Quality Assurance (QA) program previously approved by the Commission as satisfying the requirements of Appendix B to Part 50 will be accepted as satisfying the requirements of Part 72. In filing the description of the QA program required by Part 72.140(c), each licensee shall notify the NRC of it's intent to apply it's previously approved QA program to ISFSI activities. The notification shall identify the previously approved QA program by date of submittal, docket number and date of Commission approval.

Finding: This requirement was implemented. The licensee provided notification to the NRC on June 19, 2003 of their intent to apply their Part 50 Appendix B Quality Assurance Program to the construction and operation of the ISFSI. The notification contained the required information.

Documents Reviewed: Letter CNRO-2003-00026 from the Grand Gulf Station to the NRC dated June 19, 2003 (ML031820167)

Category: Quality Assurance **Topic:** Corrective Actions

Reference: 10 CFR 72.172

Requirement: The licensee shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action taken to preclude repetition. This must be documented and reported to appropriate levels of management.

Finding: This requirement was implemented. Procedure EN-LI-102 defined conditions adverse to quality as failures, malfunctions, deficiencies, deviations, defective material and equipment and non-conformances. The procedure further defined significant conditions adverse to quality as failures, malfunctions, deficiencies, deviations, defective material and equipment and non-conformances which have resulted in, or could result in, a significant degradation or challenge to nuclear safety.

Approximately 34 condition reports associated with ISFSI activities were selected for review. The conditions were characterized, assigned, corrected and closed out in accordance with Procedure EN-LI-102.

Condition reports were generated for all conditions adverse to quality. The Plant Licensing Group reviewed each condition report to characterize its risk significance. The

Condition Review Group then assigned the department(s) responsible for correcting the condition and established the due dates. Finally, the supervisor of each assigned department signed for closure of the condition report upon completion of the corrective actions. Administrative personnel then entered the completed condition report into the document management system.

In addition, all significant conditions adverse to quality were subjected to a root cause analysis. The Corrective Action Review Board (CARB), consisting of senior management personnel from each plant department, reviewed and approved the corrective actions to ensure the significant adverse condition would not recur.

Documents Reviewed: Procedure EN-LI-102, "Corrective Action Process", Revision 7

Category: Quality Assurance **Topic:** Helium Purity

Reference: CoC 1014, Tech Spec A.3.1.1.2, Table 3-2 footnote

Requirement: Helium used for backfilling the canister shall have a purity of greater than or equal to 99.995 percent.

Finding: This requirement was implemented. Grand Gulf had received 25 bottles of ultra high purity helium from the River Bend Station on August 10, 2006 under transfer ticket #52375. The helium bottles staged in the bottle racks on the 208' elevation of the Grand Gulf auxiliary building were from Airgas lot number 54-124065967-1. The Airgas Certificate of Batch Analysis dated May 19, 2006 certified that the bottles contained 99.999 percent pure helium.

Documents Reviewed: Airgas Certificate of Batch Analysis for Lot #54-124065967-1

Category: Quality Assurance **Topic:** Instruments Requiring Calibration

Reference: FSAR 1014, Table 8.1.7

Requirement: Instruments requiring calibration are listed in Table 8.1.7 of the FSAR. These include canister pressure gauges, gas and water temperature gauges, a surface pyrometer, a vacuum gauge for gas sampling and moisture monitoring instruments for Forced Helium Dehydration (FHD) operations.

Finding: This requirement was implemented. The Dew Point Monitor used for measuring canister dryness was calibrated by Echelon Laboratories on April 17, 2006 with a one year calibration interval. This instrument was required for meeting Technical Specification 3.1.1.1

The two Ashcroft digital pressure gauges and the two Ashcroft digital temperature gauges used for monitoring canister internal pressure and temperature were calibrated under the Grand Gulf Measuring and Test Equipment (M&TE) program. The Fisher Scientific stopwatch and the Fluke surface pyrometer were also calibrated under the Grand Gulf M&TE program. At the time of initial fuel loading, all instruments were within their calibration interval.

Documents Reviewed: None.

Category: Quality Assurance **Topic:** Material Receipt Inspection Checklists

Reference: FSAR 1014, Tables 8.1.8, 8.1.9, 8.1.10

Requirement: Recommended receipt inspection checklists are provided for the storage cask, transfer cask, and canister. Users shall develop site-specific receipt inspection checklists based on the recommendations.

Finding: This requirement was implemented. The receipt inspection process was defined in Procedure EN-DC-313 and formal receipt was documented using Receiving Inspection Reports (RIRs). The RIR used for the first four dry fuel storage canisters documented acceptance of the documentation packages only. Documentation included the Holtec Component Completion Record, Declaration of Conformance, Certificate of Compliance, Material Certification Records, Certified Test Mill Results, and cask fabrication records.

The physical receipt inspections for the first four canisters were performed under Work Orders 68259, 68260, 68262, and 68263 using Procedure 20-S-01-050. The inspection points contained in Procedure 20-S-01-050 were consistent with the checklists contained in the Holtec FSAR.

Documents Reviewed:
Procedure EN-MP-120, "Material Receipt", Revision 0
Work Orders 68259, 68260, 68262, and 68263, "Perform MPC Receipt, Handling and Inspection", dated July 14, 2006.
Procedure 20-S-01-050, "DFS Multi-Purpose Canister Receipt, Handling and Inspection", Revision 0

Category: Quality Assurance **Topic:** Nonconforming Material and Parts

Reference: 10 CFR 72.170

Requirement: The licensee shall establish measures to control materials, parts or components that do not conform to their requirements in order to prevent their inadvertent use or installation. These measures must include procedures for identification, documentation, segregation, disposition and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

Finding: This requirement was implemented. Procedures EN-MP-120, EN-LI-102 and EN-MP-112 were well developed and provided adequate guidance for controlling materials, parts and components that did not conform to requirements. Deficiencies were documented in condition reports. Non-conforming materials, parts and components were segregated, isolated and tagged to preclude inadvertent use or installation.

Quality Assurance personnel performed oversight observations to verify that the controls were implemented and effective. A review of approximately 30 oversight observations indicated the controls were effective.

Documents Reviewed:
Procedure EN-MP-120, "Material Receipt", Revision 0
Procedure EN-LI-102, "Corrective Action Process", Revision 7
Procedure EN-MP-112, "Shelf Life Program", Revision 0

Category: <u>Quality Assurance</u> **Topic:** <u>Transfer Cask Water Jacket Relief Valves</u>

Reference: FSAR 1014, Section 9.2.4

Requirement: The two transfer cask water jacket pressure relief valves shall be calibrated annually, or prior to the next use if the transfer cask is out of service for greater than one year. The pressure settings shall be 60 +2/-0 psig and 65 +2/-0 psig, or replaced with factory-set relief valves.

Finding: This requirement was implemented. Both relief valves were calibrated and set on November 6, 2006 under Work Orders 67549-09 and 67549-10. One relief valve was set at 60 psig and the other was set at 65 psig.

Documents Reviewed: Work Orders 67549-09 and 67549-10, "Check and Adjust Relief Valve Set Pressure", dated October 31, 2006

Category: <u>Radiation Protection</u> **Topic:** <u>ALARA Program</u>

Reference: FSAR 1014, Section 10.1.1

Requirement: Licensees using the HI-STORM 100 System will apply their existing site ALARA policies, procedures and practices to ISFSI activities to ensure that the personnel exposure requirements of 10 CFR 20 are met.

Finding: This requirement was implemented. Procedure EN-RP-110 defined the ALARA program used at all Entergy sites. The ALARA program applied to all activities involving radiological hazards and had been expanded to include ISFSI activities. The ALARA program was implemented through the EN-RP series of procedures. Procedure EN-RP-101 provided direction for access control measures. Procedure EN-RP-105 provided direction for the use of radiation work permits. Procedure EN-RP-108 provided direction for radiation area postings. Procedure 20-S-01-006 provided specific instructions for radiological monitoring requirements for each phase of the dry fuel storage operation including spent fuel handling, loading and transfer operations. All of these procedures were effectively implemented during the initial loading of spent fuel into dry storage. The ALARA goal for the first cask was established at 0.675 person-rem and the actual personnel exposure was 0.252 person-rem.

Documents Reviewed:
Procedure EN-RP-101, "Access Control for Radiologically Controlled Areas", Revision 1
Procedure EN-RP-105, "Radiation Work Permits", Revision 0
Procedure EN-RP-108, "Radiation Protection Postings", Revision 3
Procedure EN-RP-110, "ALARA Program", Revision 0
Procedure 20-S-01-006, DFS Radiological Monitoring Requirements for the HI-STORM 100", Revision 0

Category: <u>Radiation Protection</u> **Topic:** <u>Criticality Control - BWR</u>

Reference: 10 CFR 72.124(c) ; 10 CFR 50.68(b)

Requirement: A criticality monitoring system shall be maintained in each area where spent fuel is handled, which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration is not required. The SNM is considered "packaged" when the canister lid seal weld is complete.

Finding: This requirement was implemented. Criticality monitoring was provided by two Eberline EC-4 portable area radiation monitors and one AMP-100 portable area radiation monitor. An installed permanent plant radiation monitor 1D21-K625, in the spent fuel pool area, was designated as a backup monitor in the event the primary criticality monitors failed. Area radiation monitor 1D21-K625 was in service at all times.

One Eberline EC-4 monitor was located on the auxiliary building west wall and the other on the east handrail of the cask washdown pit. Both monitors were set to actuate at 200 mrem/hr above background and were equipped with audible alarms. The AMP-100 portable area radiation monitor was suspended from the fuel handling platform just above the surface of the spent fuel pool. This monitor was also equipped with an audible alarm.

Procedure 20-S-01-002, Step 6.7.4 placed the criticality monitors in service prior to lifting the loaded canister from the cask loading pool and they remained in service until the canister was dewatered in the cask washdown pit following pressure testing.

Documents Reviewed: Procedure 20-S-01-002, "DFS Cask Loading", Revision 0
Procedure 20-S-01-006, "DFS Radiological Monitoring Requirements for the HI-STORM 100", Revision 0

Category: Radiation Protection **Topic:** ISFSI Controlled Area - Accident Conditions

Reference: 10 CFR 72.106(a)(b)(c)

Requirement: For each ISFSI, a controlled area must be established. Any real individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of 5 rem total effective dose equivalent (TEDE) over 30 days. The minimum distance from the ISFSI to the nearest boundary of the controlled area must be 100 meters. The controlled area may be traversed by roads, railroads or waterways as long as arrangements are made to control traffic and to protect the public.

Finding: This requirement was implemented. The Holtec canisters have been analyzed for a number of accidents including transfer cask and storage cask handling accidents, storage cask tip over, fire, tornado, flood, earthquake, explosion, lightning, blockage of storage cask air vents, and failure of the supplemental cooling system. Section 5.1.2 of the Holtec FSAR concluded that the bounding accident would be a drop of the transfer cask resulting in loss of the water jacket. The evaluation indicated that the dose from a cask loaded with bounding fuel would be 3.08 rem in a 30 day period. This was less than the 5.0 rem limit.

The Grand Gulf site boundary was approximately 700 meters from the centerline of the ISFSI at the closest point. The plant access road located within 100 meters of the ISFSI pad was controlled by Plant Security. There were no railroads or waterways within 100 meters of the ISFSI.

Documents Reviewed: ER-GG-2003-0018-021, "Bounding Analysis of ISFSI Pad Offsite (Site Boundary) Dose Impact for Realistic, Maximum Normal, and Accident Conditions", Revision 0
Calculation XC-Q1F16-06001, "Offsite Doses from the Independent Spent Fuel Storage Installation", Revision 0

Category: Radiation Protection **Topic:** ISFSI Controlled Area - Normal Operations

Reference: CoC 1014 TS A 5.7.2; 72.104(a); 72.212(b)(2)(i)(c)

Requirement: The licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents. 10 CFR 72.104(a) specifies that during normal and anticipated occurrences the annual dose equivalent to any real individual located beyond the controlled area must not exceed 25 mrem whole body as a result of direct radiation from the ISFSI.

Finding: This requirement was implemented. The Grand Gulf Engineering Request and calculation indicated that, during normal operation and anticipated occurrences, the dose to any real individual located at or beyond the site boundary would be less than 5 mrem/year as a result of direct radiation from the ISFSI. The calculation assumed a total of 92 fully loaded HI-STORM-100S, Version B casks containing design basis fuel (47.5 GWD/MTU and 5 years decay) and continuous occupancy (8766 hours per year).

Section 3.9 of Appendix C to the 72.212 Evaluation Report stated that the worst case historical annual direct radiation dose at the site boundary was approximately 17.6 mrem. The calculated dose of 5.0 mrem/year from a fully loaded ISFSI would bring the total site boundary annual dose to 22.6 mrem/year, below the 25 mrem/year limit established in 10 CFR 72.104(a).

Documents Reviewed: ER-GG-2003-0018-021, "Bounding Analysis of ISFSI Pad Offsite (Site Boundary) Dose Impact for Realistic, Maximum Normal, and Accident Conditions", Revision 0
Calculation XC-Q1F16-06001, "Offsite Doses from the Independent Spent Fuel Storage Installation", Revision 0
ER-GG-2003-0018-000, "Entergy Nuclear 10 CFR 72.212 Evaluation Report for Independent Spent Fuel Storage Installations Utilizing the Holtec International HI-STORM 100 Cask System," Revision 5

Category: Radiation Protection **Topic:** Storage Cask Dose Rate Limits

Reference: CoC 1014, Tech Specs 5.7.3, 5.7.4, 5.7.8

Requirement: The licensee shall establish site specific surface dose rate (neutron + gamma) limits for the storage cask. The dose rate limits shall not exceed 20 mrem/hr on the top and 110 mrem/hr on the side, excluding the air vents. A minimum of 12 dose rate measurements shall be taken on the side of the storage cask in 3 sets of 4 measurements approximately 90 degrees apart at the cask mid-plane, 60 inches above mid-plane and 60 inches below mid-plane. One dose rate measurement shall be taken on the top of the lid in the center. Four dose rate measurements shall be taken approximately 90 degrees apart on top of the concrete shield halfway between the center and the outer edge. The measured dose rates shall not exceed the technical specification limits or the site specific limits, whichever are lower.

Finding: This requirement was implemented. Procedure 20-S-01-006, Attachment 1, Table 2 provided the maximum calculated surface dose rates limits for the loaded storage casks. Radiation Survey GG-0611-0193 posted the results of the post loading radiation survey on the first loaded storage cask. The dose rates are presented below.

Limit at center of the lid 4.5 mrem/hr.	Measured 0.5 mrem/hr
Limit on each quadrant of the lid 8.3 mrem/hr.	Highest measured 0.5 mrem/hr
Limit at mid-plane 41.0 mrem/hr.	Highest measured 1.2 mrem/hr
Limit at 60 inches above mid-plane 17.0 mrem/hr.	Highest measured 0.2 mrem/hr
Limit at 60 inches below mid-plane 35.1 mrem/hr.	Highest measured 0.6 mrem/hr
Limit at inlet duct 38.3 mrem/hr.	Highest measured 2.2 mrem/hr
Limit at outlet duct 17.8 mrem/hr.	Highest measured 0.9 mrem/hr.

Both the calculated dose rates and actual measured dose rates were well below the technical specification limits. The survey was performed in the manner specified by the technical specification and the first storage cask had a decay heat load of 12.6 kW.

Documents Reviewed: Procedure 20-S-01-006, "DFS Radiological Monitoring Requirements for the HI-STORM 100", Revision 0
Radiation Survey GG-0611-0193 dated 11/18/06

Category: Radiation Protection **Topic:** Transfer Cask Dose Rate Limits

Reference: CoC 1014, Tech Spec A.5.7.3, 5.7.8

Requirement: The licensee shall establish site-specific surface dose rate (neutron + gamma) limits for the transfer cask top and sides. A minimum of 4 dose rate measurements shall be taken on the side of the transfer cask approximately 90 degrees apart, at the cask mid-plane between the radial ribs of the water jacket. A minimum of 4 dose rate measurements shall be taken on the top of the transfer cask approximately 90 degrees apart, halfway between the hole in the lid and the lid outer edge. The measured dose rates shall not exceed the site-specific limits established.

Finding: This requirement was implemented. Procedure 20-S-01-006, Attachment 1, Table 1 provided the maximum calculated surface dose rates limits for the loaded transfer cask. Radiation Survey GG-0611-0176 posted the results of the post loading radiation survey on the first loaded transfer cask. The dose rates are presented below.

Limit on lid 97.0 mrem/hr.	Highest measured 2.5 mrem/hr
Limit at mid-plane 281 mrem/hr.	Highest measured 3.0 mrem/hr

Both the calculated dose rates and actual measured dose rates were well below the technical specification limits. The survey was performed in the manner specified by the technical specification and the first storage cask had a decay heat load of 12.6 kW.

Documents Reviewed: Procedure 20-S-01-006, "DFS Radiological Monitoring Requirements for the HI-STORM 100", Revision 0
Radiation Survey GG-0611-0176 dated 11/20/06

Category: Radiation Protection **Topic:** Transfer Cask Surface Contamination Limit

Reference: CoC 1014, Tech Spec A.3.2.2

Requirement: Removable contamination on the exterior surfaces of the transfer cask and accessible portions of the canister shall not exceed 1000 disintegrations per minute per 100 square centimeters (dpm/100 square centimeters) from beta and gamma sources and 20 dpm/100

	square centimeters from alpha sources. The accessible portion of the canister is the upper portion of the canister external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed.
Finding:	This requirement was implemented. Procedure 20-S-01-002, Step 6.12.6 performed the contamination survey of the accessible areas of the canister following annulus seal removal. Procedure 20-S-01-006, Step 6.7.14 and Attachment I performed the contamination survey of the exterior surface of the transfer cask and accessible areas of the canister. In both procedures, the contamination limits specified were consistent with the technical specification. Contamination Survey GG-0611-0176 indicated that all contamination levels were within technical specification limits.
Documents Reviewed:	Procedure 20-S-01-002, "DFS Cask Loading", Revision 0 Procedure 20-S-01-006, "DFS Radiological Monitoring Requirements for the HI-STORM 100", Revision 0 Contamination Survey GG-0611-0176, completed 11/17/06

Category: Records **Topic:** Cask Records Maintained by Licensee

Reference:	10 CFR 72.212(b)(8)
Requirement:	The licensee shall accurately maintain the records provided by the cask supplier for each cask that show, in addition to the information provided by the cask vendor, the name and address of the cask vendor, the listing of the spent fuel stored in the cask, and any maintenance performed on the cask. This record must include sufficient information to furnish documentary evidence that any testing and maintenance of the cask has been conducted under an NRC approved Quality Assurance plan.
Finding:	This requirement was implemented. Section 5.4 of procedure ENS-DC-160 established a Cask Document Record (CDR) for each cask. Each CDR contained the information specified by the regulation.
Documents Reviewed:	Procedure ENS-DC-160, "Dry Fuel Storage Document Control", Revision 2

Category: Records **Topic:** Copies of CoC and Related Documents

Reference:	10 CFR 72.212(b)(7)
Requirement:	The general licensee shall maintain a copy of the CoC and documents referenced in the certificate until use of the cask is discontinued. The documents referenced in the Holtec 1014, Amendment 2 certificate include the Final Safety Analysis Report, NRC Safety Evaluation Report, 72.212(b) Evaluation Report, 10 CFR 50.59 and 10 CFR 72.48 Safety Evaluations, and all related correspondence with the NRC.
Finding:	This requirement was implemented. Section 5.4 of procedure ENS-DC-160 established a Cask Document Record (CDR) for each cask. Each CDR contained the information specified by the regulation. The Certificate of Compliance and all documents referenced in the certificate were located in, and verified to be retrievable from, the licensee's document control system.

Documents Reviewed: Procedure ENS-DC-160, "Dry Fuel Storage Document Control", Revision 2

Category: Records **Topic:** Notice of Initial Loading

Reference: 10 CFR 72.212(b)(1)(i)

Requirement: The general licensee shall notify the NRC at least 90 days prior to first storage of spent fuel.

Finding: This requirement was implemented. The licensee provided notification to the NRC on May 16, 2006 of their intent to load fuel into dry storage. Initial fuel loading commenced on November 10, 2006 and the 90-day advance notice requirement was met.

Documents Reviewed: Grand Gulf letter GNRO-2006/00033 to the NRC dated May 16, 2006 (ML061380545)

Category: Safety Reviews **Topic:** Changes, Tests, and Experiments

Reference: 10 CFR 72.48(c)(1)

Requirement: A general licensee may make changes in the facility or storage cask design, make changes to procedures, and conduct tests or experiments without obtaining a Certificate of Compliance (CoC) amendment if a change in the terms, conditions or specifications of the CoC is not required AND the change, test or experiment does not; a) result in more than a minimal increase in the frequency or consequence of an accident previously analyzed, or of a malfunction of a system, structure or component (SSC) important to safety; b) create a possibility for an accident not previously analyzed, or a failure of an SSC important to safety with a different result than previously analyzed; or c) result in a exceeding a design basis limit for a fission product barrier, or departing from a method of evaluation used in establishing the design basis.

Finding: This requirement was implemented. The license had implemented Procedure EN-LI-112 that described the 10 CFR 72.48 safety review program. The program met the requirements of 10 CFR 72.48 and incorporated the guidance of NEI 96-07, Appendix B, "Guidelines for 10 CFR 72.48 Implementation." The licensee's process required that a screening for applicability of the proposed change be performed in accordance with the guidelines of NEI 96-07. Each 10 CFR 72.48 screening was then reviewed by a second qualified individual prior to processing. If the safety review screening of the proposed activity indicated that the proposed change could adversely affect a structure, system or component, the 10 CFR 72.48 evaluation was performed in accordance with the criteria found in 10 CFR 72.48(c)(2). Each 10 CFR 72.48 evaluation was required to have a number assigned and to be reviewed by the On-Site Safety Review Committee. At the time of the inspection, the licensee had performed several 10 CFR 72.48 screenings but no evaluations.

Documents Reviewed: Procedure EN-LI-112, "10 CFR 72.48 Review Program," Revision 2
10 CFR 72.48 Review Form for ER-GG-0018-044, "150-Ton Cask Handling Crane Radio Control," Revision 0

Category: Training **Topic:** Approved Training Program

Reference: 10 CFR 72.44(b)(4); 72.190; 72.194

Requirement: The licensee shall have an NRC approved training program in effect that covers the training and certification of personnel that meets the requirements of Subpart I before the licensee receives spent fuel and/or reactor related Greater Than Class C waste at the ISFSI. Subpart I references to Part 72.190 and 72.194 for a general license. Part 72.190 specifies that only trained and certified personnel (or persons under the direct visual supervision of a certified individual) may operate equipment and controls identified as important to safety in the Safety Analysis Report and in the license. Part 72.194 specifies that the physical condition of certified personnel must not be such as might cause operational errors that could endanger other in-plant personnel or the public health and safety.

Finding: This requirement was implemented. The Dry Fuel Storage training program was described in GPCS-DFS-INITL. The training was developed and implemented using the Systematic Approach to Training (SAT) process, as defined in Procedure EN-TQ-201, and was approved under the licensee's 10 CFR Part 50 training program.

Plant Operations personnel received overview training on the dry fuel storage project and specific training in the dry fuel storage cask loading sequence and associated technical specifications. Radiation Protection personnel received overview training on the dry fuel storage project and specific training on the radiological coverage requirements. Fuel Services Technicians and supervisors received overview training on the dry fuel storage systems and components.

Fuel Services Technicians and supervisors received specific classroom training and On-The-Job (OJT) training in: a) canister pressure testing, blowdown, drying and helium backfill using the RVOAs, hydrostatic test rig and FHD system; b) canister cooling using the supplemental cooling system and water recirculation systems; c) heavy lift operations from the cask loading pool to the ISFSI pad using the rigging plan, lift yoke, mating device, low profile transporter, and vertical cask transporter; and d) receipt inspection of the cask system components.

Classroom attendance and successful completion of examinations were documented in the student training histories located in the licensee's ONTRACK computer database. Successful completion of OJT was documented on the Fuel Services Technician (FST) Qualification Cards.

The physical condition of dry fuel storage personnel was evaluated under the licensee's medical program, as described in Procedure EN-NS-112. Section 5.6 of Procedure EN-NS-112 included the types of physical examinations provided. All 6 Fuel Services Technicians had received respirator and crane operator physicals within the past year. The scope of these physical examinations was adequate to ensure that the physical condition of the certified personnel met the requirements of the regulation.

Documents Reviewed: Training Program GPCS-DFS-INITL, "Grand Gulf Nuclear Station Dry Fuel Storage Training", Revision 0
Procedure EN-TQ-201, "Systematic Approach to Training Process", Revision 1
Procedure EN-NS-112, "Medical Program", Revision 2
Qualification Card GQC-DFS-FST01, "Fuel Services Technician (FST) Qualification Card", Revision 2

Lesson Plans:
GLP-RPCT-DYFLST, "GGNS Dry Fuel Storage Project Overview", Revision 2
GLP-RPCT-DFUP, "GGNS Dry Fuel Storage Project Update", Revision 0
GLP-RPCT-DFSCV, "Holtec DFS Radiological Coverage Requirements", Revision 2
GLP-OPS-DYFLST, "Dry Fuel Storage Overview", Revision 2
GLP-OPS-DFS01, "Dry Fuel Storage Cask Loading", Revision 0
GLP-OPS-DYFTS, "Dry Cask Storage Tech Specs", Revision 0
GLP-DFS-STORM, "HI-STORM 100 System Overview", Revision 0
GLP-DFS-FHD, "Forced Helium Dehydration", Revision 0
GFAM-DFS-RIG, "DFS Rigging Plan", Revision 0
GLP-DFS AUXMOVE, "HI-TRAC Movement and MPC Transfer", Revision 0
GLP-DFS-MPCPREP, "HI-TRAC Preparation for MPC Fuel Loading and Processing", Revision 0
GLP-DFS-RECEIPT, "High Storm 100 System Receipt Inspections", Revision 0
GLP-DFS-SFCST, "Spent Fuel Cask Site Transportation", Revision 0 (FST and Supervision)
GLP-DFS-VCT, "4-Point Lift Systems Cask Transporter Operation", Revision 0

Category: Training **Topic:** Dry Run Exercise

Reference: CoC 1014, Condition 10

Requirement: A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM 100 System shall be conducted by the licensee prior to first use of the system to load spent fuel assemblies. The dry run shall include the operations specified in CoC Condition 10.

Finding: This requirement was implemented. Condition 10 required pre-operational testing of the HI-STORM 100 Cask System prior to loading spent fuel assemblies. The pre-operational testing program required testing of 11 specific operations lettered a through k.

Operation 10.a moved the transfer cask from the cask washdown pit to the cask loading pool. Operation 10.b prepared the cask system for fuel loading. These operations were conducted during pre-operational testing on October 23-28, 2006.

Operation 10.c verified the fuel selection and verification process was capable of identifying fuel assemblies meeting the specifications contained in the Certificate of Compliance. The fuel selection and verification process was evaluated during the programs inspection on October 16-19, 2006.

Operation 10.d loaded the dummy fuel assembly into the canister. Operation 10.e installed the canister lid underwater and moved the transfer cask from the cask pool back to the cask washdown pit. These operations were conducted during pre-operational testing on October 23-28, 2006.

Operation 10.f required spent fuel canister welding, NDE inspections, pressure testing, draining, moisture removal and helium backfilling. Canister welding and NDE inspection operations were performed during pre-operational testing on March 21-24, 2006 and documented in Inspection Report 50-416/06-09; 72-050/06-01 (ML061000731). Canister pressure testing, draining, moisture removal and helium backfilling were performed during pre-operational testing on September 13-15, 2006 and documented in Inspection Report 50-416/06-13; 72-050/06-02 (ML062760476)

Operation 10.g required testing of the transfer cask supplemental cooling system (SCS). Operation of the SCS was performed during pre-operational testing on September 13-15, 2006 and documented in Inspection Report 50-416/06-13; 72-050/06-02 (ML062760476)

Operation 10.h required testing of the transfer cask horizontal handling equipment. Horizontal operations are not conducted at the Grand Gulf Station and this operation was not applicable.

Operation 10.i transferred the canister from the transfer cask into the storage cask. Operation 10.j placed the storage cask at the ISFSI. These operations were conducted during the pre-operational testing on October 23-28, 2006.

Operation 10.k tested unloading of the cask system, including cooling fuel assemblies, flooding the canister cavity, and removing the canister lid welds. Cooling of the fuel assemblies and flooding the canister were performed during pre-operational testing on September 13-15, 2006 and documented in Inspection Report 50-416/06-13; 72-050/06-02 (ML062760476). Removing canister lid welds was demonstrated during pre-operational testing on March 21-24, 2006 and documented in Inspection Report 50-416/06-09; 72-050/06-01 (ML061000731).

Documents Reviewed: None.

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